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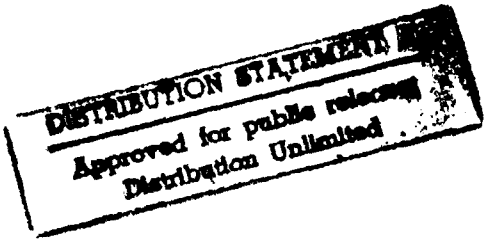

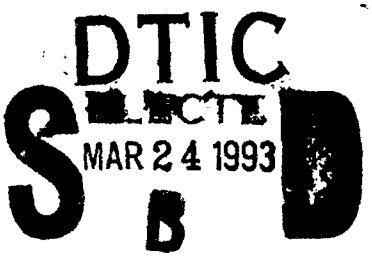

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ABSTRACT

The addition of a radiography through-tube at the SLOWPOKE II reactor at Royal Military College, Kingston produced some inexplicably high nuclear radiation fields. As a result, DREO was called upon to make detailed dosimetric and spectroscopic measurements at various locations around the facility. This marked the first field trial of the new DREO mobile nulcear laboratory. The results confirmed the preliminary dosimetry measurements, while adding greatly in detail and accuracy. The fields are viewed as, in many areas, too high to allow full-power reactor operation.

RESUMÉ

Suite à l'ajout d'un nouveau tube de radiographie au réacteur SLOWPOKE II du collège militaire de Kingston, un niveau de radiation nucléaire anormalement élevé a été observé. Ceci a permis au CRDO d'utiliser pour la première fois son nouveau laboratoire nucléaire mobile. Des mesures dosimétriques et spectroscopiques détaillées ont été faites autour du réacteur. Les résultats ont confirmés les mesures dosimétriques préliminaires, en étant toutefois beaucoup plus précises et détaillées. Les radiations sont considérées en plusieur endroits beaucoup trop intenses pour permettre l'opération du réacteur à pleine puissance.

EXECUTIVE SUMMARY

The new DREO mobile nuclear laboratory was called upon to make detailed dosimetric and spectroscopic measurement of the nuclear radiation fields around the radiography facility of the SLOWPOKE II reactor at Royal Military College, Kingston. The results show that in many areas these fields are too high to allow full-power operation.

TABLE OF CONTENTS

LIST OF FIGURES

1.0 INTRODUCTION

The SLOWPOKE II reactor facility at Royal Military College (RMC) (1), Kingston, has provided the Department of National Defence in general, and Defence Research Establishment Ottawa (DREO) in particular, with a venue for conducting a variety of radiation experiments including dosimetry (2), neutron activation analysis (3) and Transient Radiation Effects on Electronics (TREE) (4). Recently a neutron radiography facility has been constructed at the SLOWPOKE (5) consisting of a vertical through-tube pointed slightly to one side of the core, hopefully allowing a collimated thermal neutron beam. Preliminary measurements by RMC staff indicated that the radiation fields around the radiography facility were higher than expected. As a result, DREO was requested to perform detailed neutron and gamma-ray dosimetric and spectroscopy measurements in order to determine the cause of and potential shielding against these excess fields.

This request dove-tailed very nicely into DREO's own experimental plans. The new DREO Mobile Nuclear Laboratory (MNL) (6) was ready for its maiden field test, and this provided a perfect opportunity to assess its capabilities.

The measurements were conducted over the period 17-20 June, 1991, and the results presented here show conclusively the need for improved shielding in the radiography area.

2.0 EXPERIMENTAL

2.1 The RMC Radiography Facility

Figure 1 shows the RMC through-tube Radiography facility. Note that it is oriented at a slight angle from the vertical, with its base being slightly outside of the reactor container (approx 30cm diameter). The beam stop, or "box" is located at ground level in the SLOWPOKE room.

3.0 RADIATION DETECTION EQUIPMENT

The equipment used here represents a subset of that available in the new MNL, as presented in a companion report (6).

3.1 Dosimetry

Gamma-ray dosimetry was achieved using a hand-held meter and two types of thermoluminescent dosimeters (TLDs). The hand-held meter was the Eberline Model ASP-1 outfitted with a probe. The detector is capable of measuring dose rates from

background (≤ 10 μRad/h) up to a few Rads/h. It is DREO's principle early radiation indicator. The errors on the dose rates reported here depend, of course, on meter stability during reading, but should be of the order of ± 10 - 20%.

Two types of CaF_2:Mn TLDs were employed:

(a) Small rods, 50 in total, 1 mm x 1 mm x 6 mm, capable of measuring down to ~ 10 mRad;

(b) Large squares, 30 in total, .25" x .25" x 0.035", capable to measuring down to ~ 2 mRad.

The type used was dependent upon the integrated dose expected.

The small rods were individually calibrated in 100 mR (^{60}Co) fields at DREO and errors in dose are typically of the order of ±5%.

For this work the large squares had not been individually calibrated, but rather the batch had been calibrated (at the same position) in a 100 mR (^{60}Co) field. The results yielded a mean and standard deviation of (30.14 ± 2.98) nC/100 mR. For all results reported here, 3 or 4 squares were used. Referring to ASTM (7), the corresponding d-value for 3-4 chips is 1.7 - 1.2.

Here d is the difference between a group mean and the overall mean, i.e.,

$$d = \frac{|\, m_A - m_B \,|}{\sqrt{2\sigma^2}} = \frac{\delta}{\sigma\sqrt{2}} \qquad (1)$$

Solving for δ in our case yields approximately 15% to 20% accuracy for the low dose measurements reported here. It should be noted that with individual calibration and/or use of newly instituted Aluminum Oxide TLDs, DREO's low dose accuracy has improved to ±5%, with a lower limit ~ 0.2 mRad.

One should also note that all gamma-ray results quoted here are in KERMA. (i.e. mRad). True conversion between measured TLD charge and Rads requires complete knowledge of the gamma-ray energy spectrum via the relationship

$$K_\gamma = \int \phi_\gamma(E)\, KF_\gamma(E)\, dE \qquad (2)$$

where

K_γ = total gamma-ray kerma (Rads)

ϕ_γ = gamma-ray fluence (photons/MeV cm^{-2})

KF_γ = gamma-ray energy-dependent kerma factor (Rad-cm^{-2})

and also requires a knowledge of the relationship between exposure (R) and Kerma (Rads) in the ^{60}Co calibration field via

$$D_{TLD}(Rads) = \frac{(\mu_{en}/\rho)_{TLD}}{(\mu_{en}/\rho)_{air}} (.869X(R)) \qquad (3)$$

where

D_{TLD} = absorbed dose (or kerma) delivered to the TLD (Rads);

$(\frac{\mu_{en}}{\rho})$ = mass energy-absorption coefficient in TLD/air (g-cm^{-2});

and X = exposure in air (Roentgen).

To determine the required conversion to tissue kerma:

$$K_\gamma(Tissue) = \frac{(\mu_{en}/\rho)\,tissue}{(\mu_{en}/\rho)_{TLD}}\, D_{TLD} \qquad (4)$$

Solving at ^{60}Co energies for CaF_2:Mn TLDs yields

$$K_\gamma(Tissue) = 0.96X \qquad (5)$$

Fig. (2) (taken from (3)) shows that the energy-dependent response of TLDs (when wrapped in a tin energy-compensation shield) only varies significantly from unity for very low energies. For normal fission/neutron capture spectra, the percentage of of total kerma in this low energy region is small and the detector will only slightly over-respond - tending to compensate for the under-response from equation (5). Thus, within experimental limits here, a 1-to-1 conversion between R and Rads (and Rem, if desired) may be used, to within a few percent.

Neutron dosimetry here was performed using bubble detectors (8). The detectors used were of three different sensitivities: 0.56 bubbles/mrem (total of 7 detectors), 8 bubbles/mrem (1 detector only), and 56 bubbles/mrem (1 detector only). Thus a considerable dynamic range was available.

The bubble detectors are calibrated in a PuBe neutron field (average energy ~ 3 MeV). To determine their validity of use in a fission/degraded fission field the energy response must be known. Many DREO Van de Graaff experiments have yielded the energy response shown in fig. (3). (eg. ref (9))

The actual response of the detector to an arbitrary neutron energy spectrum $\phi_n(E)$ is then

$$N_F = \int \phi_n(E)\, R(E)\, dE \qquad (6)$$

where N_F = observed number of bubbles

and R = energy-dependent bubble response (bubbles/fluence)

The dose equivalent one would achieve using the manufacturer's calibration is

$$[DE]_{cal} = \left(\frac{No.bubbles}{calibrationfactor}\right) = \frac{N_F}{\left[\frac{\int \phi_p(E)\, R(E)\, dE}{\int \phi_p(E)\, DE(E)\, dE}\right]} \qquad (7)$$

where DE = energy-dependent fluence to dose-equivalent conversion factor (Rem-cm^2)
ϕ_p = PuBe neutron energy spectrum (cm^{-2}).

This may be compared with the actual Dose Equivalent in the ϕ_n field.

$$[DE]_{act} = \int \phi_n (E) \quad DE(E)\, dE \qquad (8)$$

This comparison is made in Table (1) for the 'actual' and 'calibrated/measured' responses of the bubble detector in a pure fission (Watt Spectrum) and pure fission convolved with a 1/E distribution field (neither spectrum normalized).

TABLE 1

ACTUAL(act) AND BUBBLE DETECTOR(cal) MEASURED NEUTRON DOSE EQUIVALENT RATES IN TWO DIFFERENT ENERGY SPECTRA

Field	$[DE]_{act}$ (Rem/h)	$[DE]_{cal}$ (Rem/h)
Pure Fission	6.01×10^{-4}	6.35×10^{-4}
Fission Convolved with 1/E	3.54×10^{-4}	4.34×10^{-4}

Note that except in the case of a very soft spectrum, the bubble detectors give adequate (within 10%) results. Since the anticipated spectra here (verified later by measurements) were between the two above cases (in terms of spectral hardness) and in view of the fact that the statistics of the bubbles counted were ~ 8-10%, the calculated dose equivalents were used as is. A slight overestimate may be expected based on the above.

Finally it should be noted that the neutron results are reported in Rem (dose equivalent) as opposed to the NATO standard Rad (Kerma). This was done primarily to facilitate comparison between bubble and spectroscopic results - however, in view of the fact that these experiments can be construed as a Health Physics exercise, reporting in Rem has validity.

Conversion to Rad would simply involve division by a quality factor Q, defined as

$$Q = \frac{\int DE(E)\ \phi(E)\, dE}{\int K(E)\ \phi(E)\, dE} \qquad (9)$$

This value does not vary greatly with spectral change - ranging from 10.8 for PuBe to 12.6 for a pure fission spectrum to 13.5 for the fission spectrum convolved with 1/E.

3.2 Spectroscopy

Gamma-ray spectroscopy was accomplished using a BGO scintillator (9). The detector, coupled to a dual-gain amplifier system is usuable over the energy range 0.1 - 12.0 MeV (thus detecting almost all neutron capture gamma rays) with reasonable (few hundred keV) resolution.

Neutron spectroscopy was achieved using a coupled NE213/BF_3 system. The NE213 measures directly the spectrum from 0.6 - 16 MeV (using a dual-gain system). The paired BF_3 (cadmium-covered and bare) then measure the epithermal fluence (at 0.5 eV) and a power fit of the form

$$\phi(E) = A\ E^p \qquad (10)$$

joins the epithermal to 700 keV fluence. Note that for a true 1/E spectrum p = -1.

This method of extrapolation clearly does not allow for any structure in the thermal to 0.8 MeV region and relies heavily on accurate measurement of thermal and NE213 near-threshold fluences. This has, in the past (9), resulted in inaccuracies and DREO is in the process of changing to a ROSPEC spectrometer (10) system, which should be implemented in 1992. This system makes direct spectral measurements in the 60 keV - 4 MeV range, and as such is ideally suited for fission/degraded fission spectra.

4.0 RESULTS

4.1 In-Box Dosimetry and Spectroscopy

Following an initial investigation with the ASP-1, the small TLD rods and most insensitive bubble detectors were used to delineate the gamma-ray and neutron doses inside the radiography box, both with and without the lead sheet shielding.

Table (2) gives the measured ASP-1 dose rates (scaled to full power) as a function of distance from the entrance sliding doors (with the Pb sheets in place).

TABLE 2

MEASURED ASP-1 DOSE RATES (SCALED TO FULL POWER) IN RADIOGRAPHY BOX (Pb SHEETS IN PLACE)

Distance from Entrance (cm)	Dose Rate (Rad/h)
0	0.45
10	0.110
20	0.220
30	0.620
40	2.00
50	1.60
60	1.60
70	1.60
80	1.60
100	2.40

The TLDs and bubble detectors were placed on the 'oven rack' experimental tray and the dose distributions measured. Fig. (4) gives the results (scaled to full power).

As can be seen the neutron dose equivalent is quite constant over the radiography area, at (5.27 $\pm$ 0.43) Rem/h. Note that this is only the 'fast' (> 100 keV) neutron contribution. An estimated value for the fluence may be obtained by evaluation of the average fluence to dose equivalent conversion factor <DE> for fission/degraded fission spectra defined as:

$$\langle DE \rangle = \frac{\int DE(E)\,\phi(E)\,dE}{\int \phi(E)\,dE} \qquad (11)$$

This value ranges from 3.68 X 10^{-8} Rem - cm^2 to 3.13 X 10^{-8} Rem-cm^2 to 1.64 X 10^{-8} Rem-cm^2 for PuBe, pure fission and 1/E convolved fission spectra respectively.

Since the spectrum has not been accurately measured here, a value of 2.5×10^{-8} Rem - cm^2 is arbitrarily assumed, giving a full-power fast-neutron flux of $(6 \times 10^4 \pm 2 \times 10^4)$ n/cm^2 - s.

The gamma-ray dosimetry results show variation across the radiography area of up to factors of 2 with the south side doses being greater. This is caused by the angle of the radiography through tube and non-uniform Pb shielding thickness. Note that the ASP-1 and TLD measurements are in reasonable agreement, considering that the ASP may have been experiencing dead-time effects on its highest range.

The effect of removing the lead sheets on neutron and gamma-ray doses is shown in Fig (5). As expected, the fast neutron dose-equivalent rate does not change within errors, being now (5.25 ± 0.43) Rem/h. The gamma ray doses, as expected, have gone up and are somewhat more uniform than before - although the south side still experiences more dose. The increase in dose rate of $1.83 \pm .54$ may be compared with that expected using the mass absorption coefficient for lead for 1 MeV photons (a reasonable average for fission environment gamma rays). The Pb sheets had total thickness varying from 1 cm to 1.5 cm. Using a value of μ/ρ (1 MeV) = 0.0377 cm^2/g (11), then the expected increase in dose rate would be 1.53 - 1.90 in reasonable agreement with the measured.

In-box spectroscopy was severely limited by the high dose rates and consequent dead-time problems experienced by BGO, NE213 and BF, detectors. As a result, no spectroscopic integral values (kerma or dose-equivalent) will be reported, however the BGO gamma-ray energy spectrum from the centre of the radiography area (Fig (6)) is informative to view in a qualitative way.

Note the presence of a number of peaks in the spectrum. Some, such as the 511 keV positron annihilation peak and the 2.2 MeV thermal neutron capture line in hydrogen are to be expected.

The other lines however require discussion. The dominance of the line at ~ 7.5 MeV and other smaller peaks is clearly due to neutron capture gamma rays. Some proposed source elements for these lines are nitrogen (from air in the beam tube) and iron, lead and aluminum from various structural materials. Tables (3) lists the important lines from each element (12).

TABLE 3

INTENSE ($I\gamma > 10\%$) NEUTRON-CAPTURE GAMMA RAY TRANSITION ENERGIES FROM VARIOUS ELEMENTS

Element	$E\gamma$(Mev)	I (per 100 gamma ray captures)
Nitrogen	10.8	14
Nitrogen	6.3	18
Nitrogen	5.6	11
Nitrogen	5.5	21
Nitrogen	5.3	55
Nitrogen	4.5	16
Nitrogen	3.7	23
Nitrogen	1.9	21
Aluminum	7.7	32
Aluminum	4.7	16
Aluminum	4.1	15
Aluminum	3.5	14
Aluminum	3.0	20
Aluminum	2.2	13
Aluminum	1.8	100
Iron	7.6	32
Iron	6.0	16
Iron	1.7	15
Lead	7.4	95

The fact that all three metals have strong lines within BGO resolution of 7.5 MeV, coupled with the lack of an extremely strong complementary line makes absolute elemental determination here difficult. However, a large number of smaller peaks roughly correspond to nitrogen energies. Fig (7) shows an in-box BGO spectrum acquired near the sliding doors. Note that there is some evidence for the 9.28 MeV iron capture line here - however not enough to be totally convincing that iron is the only origin of the 7.5 MeV peak. A high resolution (Ge - detector) spectroscopy system would enable positive identification. Although inside the box the neutron fluence rate would deleteriously effect such a detector, as will be seen later there are enough transmitted gamma rays above the box to allow low-rate measurements.

4.2 Leakage Dose and Spectra

Dosimetric and spectral measurements were made of the neutrons and gamma-rays leaking from the top surface of the radiography box. Three sets of measurements were made; (a) with the lead sheets in place, (b) with the lead sheets removed; and (c) with the helium replacing air in the radiography beam tube. Integral results (from the energy spectrum convolved with the appropriate conversion factors) expressed in mrad/h and mrem/h for the centre of the box are given in table (4). Note the good agreement in all cases between different detectors.

TABLE 4

MEASURED NEUTRON AND GAMMA-RAY DOSE RATES AT THE CENTRE OF TOP OF RADIOGRAPHY BEAM STOP

GAMMA RAYS

Case 1: Lead Sheets In	
Detector	Dose Rate (mRad/h)
BGO	4.7 ± 0.3
ASP	5.0 ± 0.5

Gamma Rays (continued)

Case 2: Lead Sheets Out	
Detector	Dose Rate (mRad/h)
BGO	4.6 ± 0.4
ASP	5.0 ± 0.5
TLD (large squares)	5.2 ± 1.0

Case 3: Lead Sheets Out (He in Tube)	
Detector	Dose Rate (mRad/h)
BGO	5.1 ± 0.3
ASP	5.0 ± 0.5
TLD (large squares)	4.8 ± 0.8

NEUTRONS

Case 1: Lead Sheets In	
Detector	Dose - Equivalent Rate (mRem/h)
NE213/BF_3	120 ± 10
Bubble (0.56 bubbles/mrem sensitivity)	180 ± 40

Neutrons (continued)

Case 2: Lead Sheets Out	
Detector	Dose - Equivalent Rate (mRem/h)
NE213/BF_3	148 ± 10
Bubble (0.56 bubbles/mrem sensitivity)	165 ± 40

Case 3: Lead Sheets Out/He in	
Detector	Dose - Equivalent Rate (mRem/h)
NE213/BF_3	235 ± 30
Bubble (0.56 bubbles/mrem sensitivity)	239 ± 30

In order to obtain an estimate of the leakage gradients across the box, TLDs and bubble detectors were placed at the northwest edge, centre and southeast edge locations (Pb sheets out). The results are summarized in Table (5).

TABLE 5

DOSIMETRIC SURVEY OF TOP OF BOX

Location	NW	Centre	SE
Gamma Ray (mrad/h)	5.5 ± 0.7	5.2 ± 1.0	20.5 ± 8.6
Neutron (mrem/h)	73 ± 15	165 ± 16	92 ± 20

The results for neutrons showed the expected gradient due to side shielding, however the gamma rays proved most peculiar. The anomalously high and widely spread dosimetric measurements in the SE corner lead to further investigation. Using the ASP-1 a 'hot spot' was discovered near the SE corner of the box, giving dose rates ~ 10mR/h. Following this a detailed mapping of the hot spot area was conducted with the small TLDs. The results of this mapping appear in fig (8). Clearly a small crack in the shielding material is allowing significant streaming of gamma rays. More (radiographic) measurements would be beneficial in identifying other 'hot spots'.

Figs (9) to (12) show some of the measured spectral information. Fig (9) compares the in-box to on-box gamma-ray spectra, normalized at the 7.5 MeV peak. Note that the shielding material tends to degrade the spectrum, adding to the continuum from ~ 2 MeV to 7 MeV, while below this energy absorption takes place. Fig (10) compares the gamma-ray energy spectra with the Pb sheets in and out. The only discernable difference is a slight increase in the low energy contribution without the Pb. This effect of course, is extremely small when compared to the total box shielding effect.

Fig (11) compares the gamma-ray spectrum at the hot spot compared to the centre of the box. The more well defined peaks and relatively smaller low energy contribution are to be expected.

The neutron energy spectrum for the measurement with Pb sheets in place appears in fig (12). As expected, no real structure is observed with the possible exception of a peak around 2.4 MeV. This peak is at least partially a detector artifact but may also result from neutron transmission through a 'neutron window' in oxygen (13) which has been observed by DREO in air-transported spectra (9).

The low-energy spectral shape is very interesting. For all of the neutron spectra measured here the power fit method yielded a value of p ~ -0.8, i.e. the spectrum is significantly harder than 1/E. In order to ascertain the validity of such numbers, some simple calculations were made using the DREO-developed version of ANISN called "STREAM" (14). The neutron spectra out to 2 m in water from a U-235 fission source were calculated. Some of these results are plotted in fig (15). Note that out to 1 m a true 1/E spectrum (which would be represented by a horizontal line on these plots) is not obtained. In fact a power fit for these spectra from 0.5 eV to 800 keV yields a p-value of ~ -0.8. As the radiography facility points to ~ 50 cm from the core centre, the results may be valid. However, again, these measurements must be repeated once a ROSPEC system becomes available.

4.3 Dose and Spectra in Area Around Pool

Measurements were made in the walkway area around the SLOWPOKE pool, at a height of 1 m. This would be representative of possible personnel exposure rates while radiography measurements were being made.

The spectral measurements here yielded a BGO gamma-ray kerma (full power) of 1.38 mRad/h, and an NE213 neutron dose-equivalent (full power) of 15.9 mRem/h, indicating neutron and gamma-ray reduction factors of 9.3 and 3.4 respectively from the top of the box.

Fig (13) compares the gamma-ray spectra from the walkway area with that of the top of the box. The expected softening and low-energy degradation are apparent.

Fig (14) compares the neutron spectra for the two above cases. Note that the walkway spectra is considerably softer.

4.4 Neutron Dosimetry at Other Remote Locations

Neutron dose-equivalent rates were also measured, using the most sensitive bubble detector, in (i) the reactor control room and (ii) the 2nd floor lab above the plug for the pool (at floor level). The dose-equivalent rates (normalized to full power) were (i) 1.29 mrem/h and (ii) 0.4 mrem/h.

5.0 CONCLUSIONS

The measurements reported here show significant fast neutron and gamma-ray fields in the radiography area, and higher than expected fields in personnel areas. It is recommended that additional shielding be installed around the facility. Following this, follow-up measurements should be conducted by DREO to verify the integrity of the shielding. DREO should be able to use its ROSPEC neutron and field-Ge gamma-ray spectrometers which would greatly add to the information obtained.

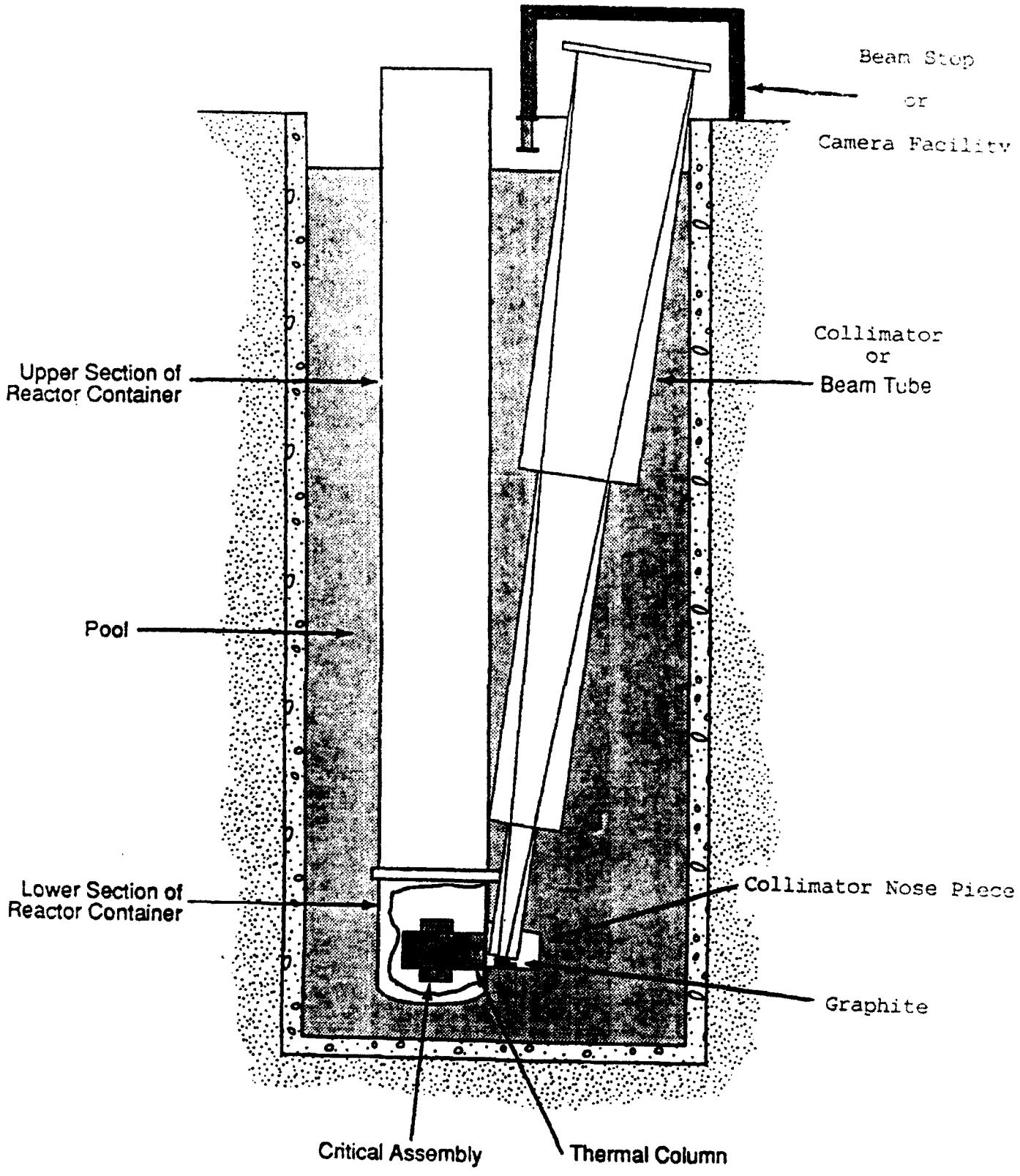

Figure 1: Cutaway diagram of the RMC neutron radiography facility. Note the angle the beam tube makes with the core.

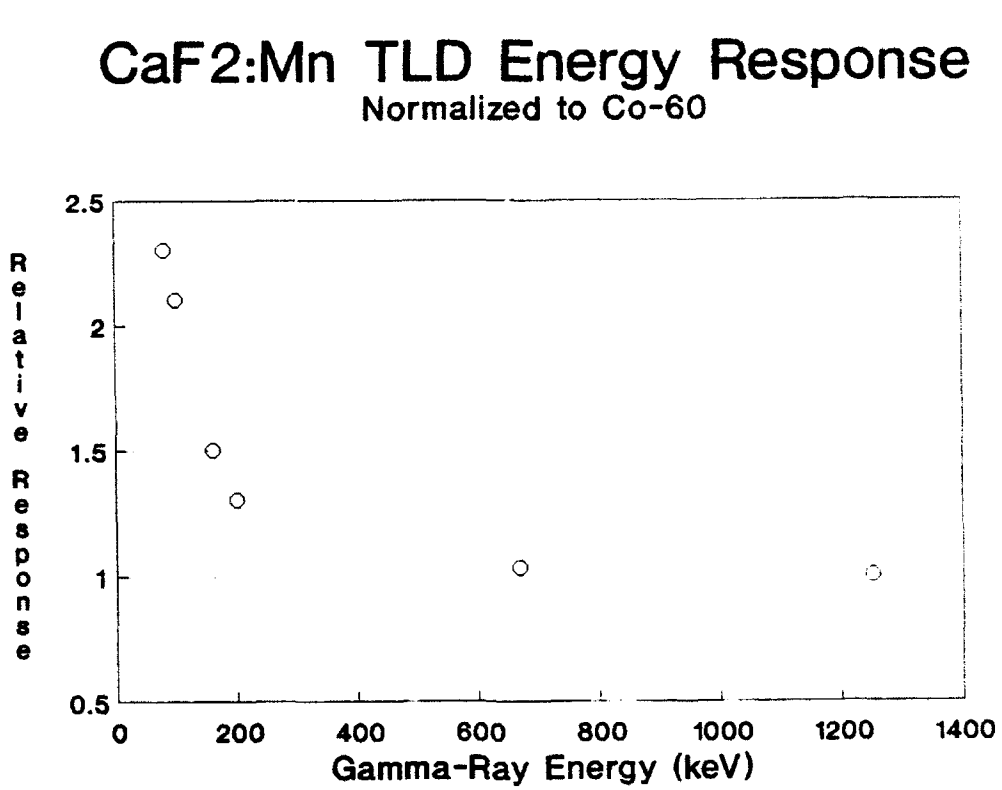

Figure 2: Measured energy dependent response of CaF_2:Mn TLDs in tin shields.

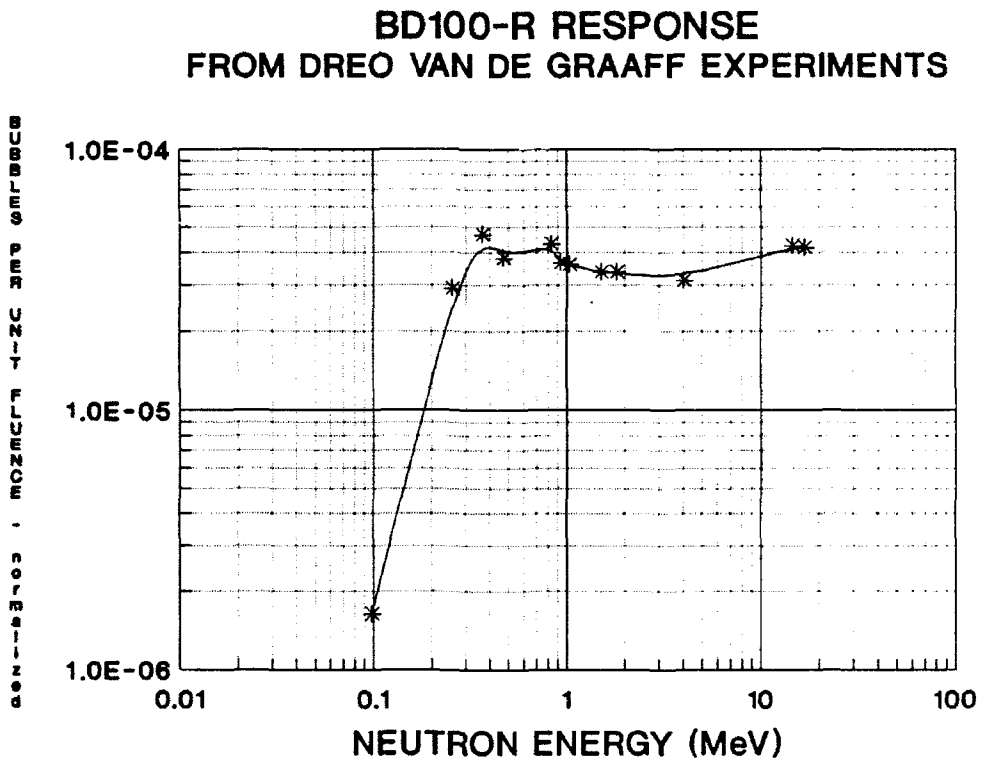

Figure 3: Measured energy dependent response of bubble detectors.

N
↑

(Scale (cm))

0		24		36
(1.50)		(2.33) [5.62]		(1.87)
		[5.25]		
[5.36]		(2.52)		(4.28)
	(2.37)	[5.94]	[4.44]	[5.36]
		[5.62]		
(3.44)		(4.69)		(4.38)

DOORS

(gamma-ray dose rate is Rad/h)

[neutron dose equivalent rate in Rem/h]

Figure 4: Full-power neutron dose-equivalent and gamma-ray dose rates in the radiography 'box' (on oven rack) with lead sheets in place.

N
↑

(Scale (cm))

0		24		36
(3.60)		(6.15) [5.46]		(3.21)
		(5.73) [5.67]		
	(4.97)	(5.59)	(7.55)	(5.55)
	[5.09]	[4.39]	[5.51]	[5.14]
		(6.06) [5.46]		
	(5.73)	(5.32)		(5.72)

DOORS

(gamma-ray dose rates in Rad/h)
[neutron dose equivalent rates in Rem/h]

Figure 5: Full-power neutron dose-equivalent and gamma-ray dose rates in radiography 'box' (on oven rack) with lead sheets removed.

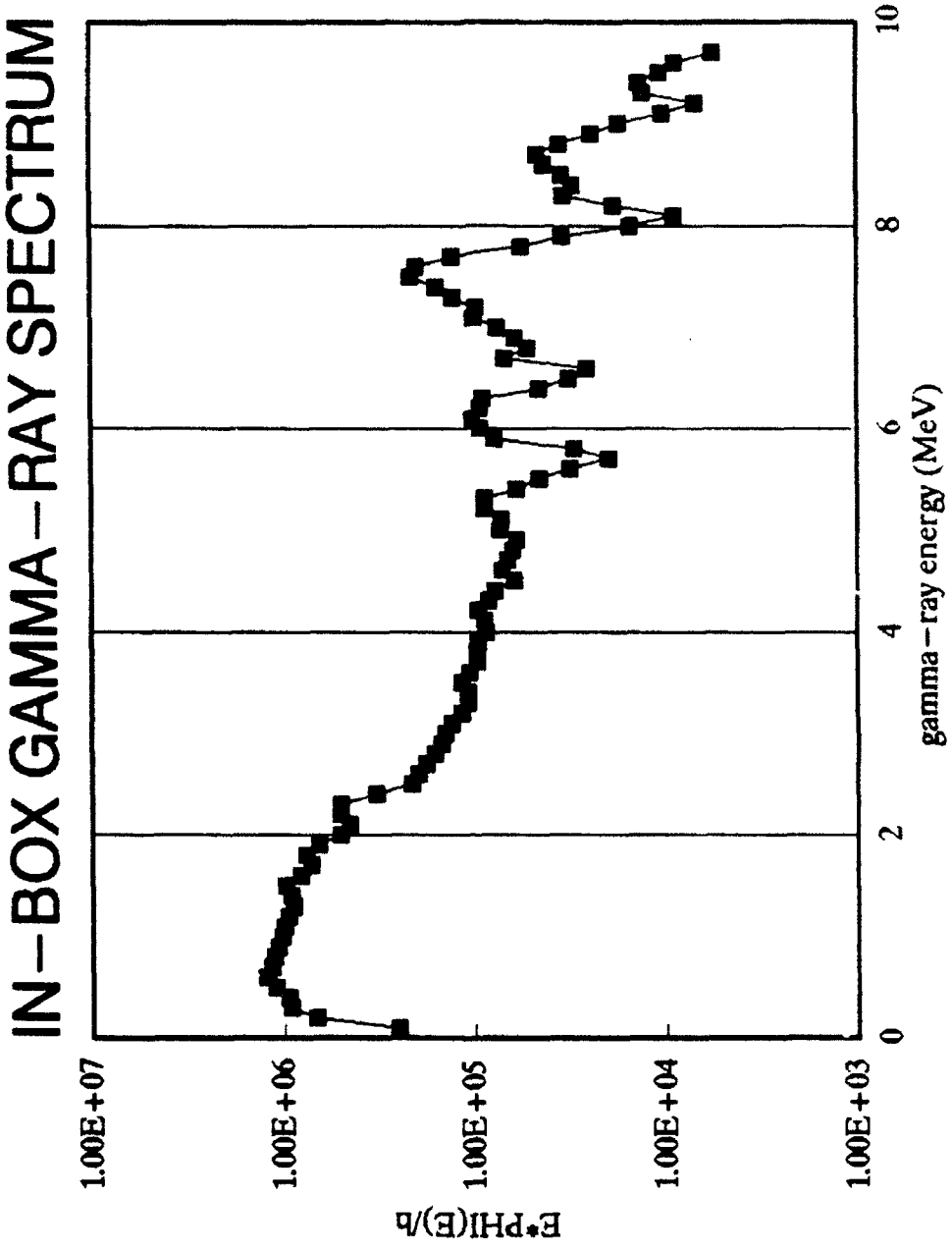

Figure 6: BGO gamma-ray energy spectrum in box, at 1/20th reactor power, normalized to 1 hour.

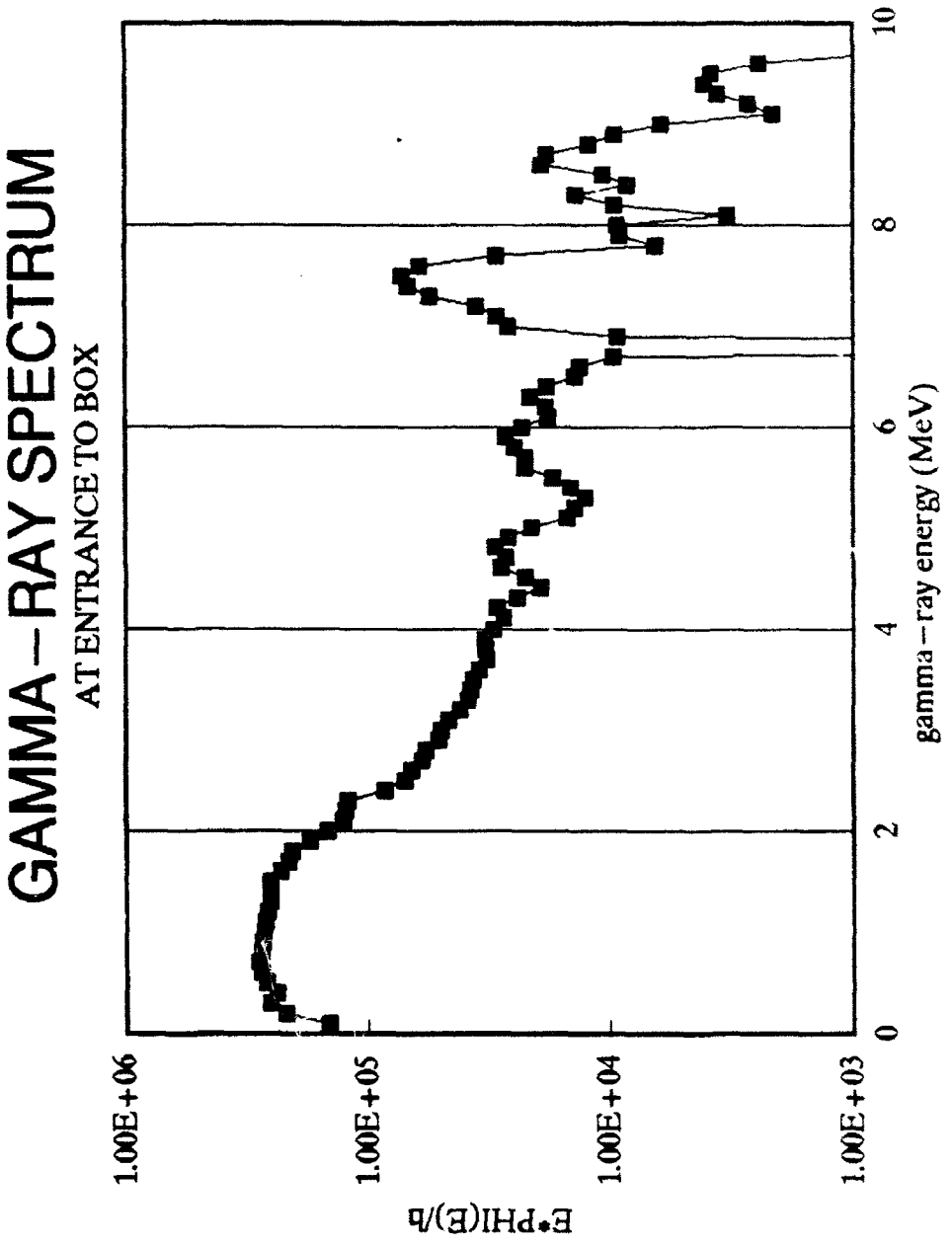

Figure 7: BGO gamma-ray energy spectrum at edge of box, at 1/20th reactor power, normalized to 1 hour.

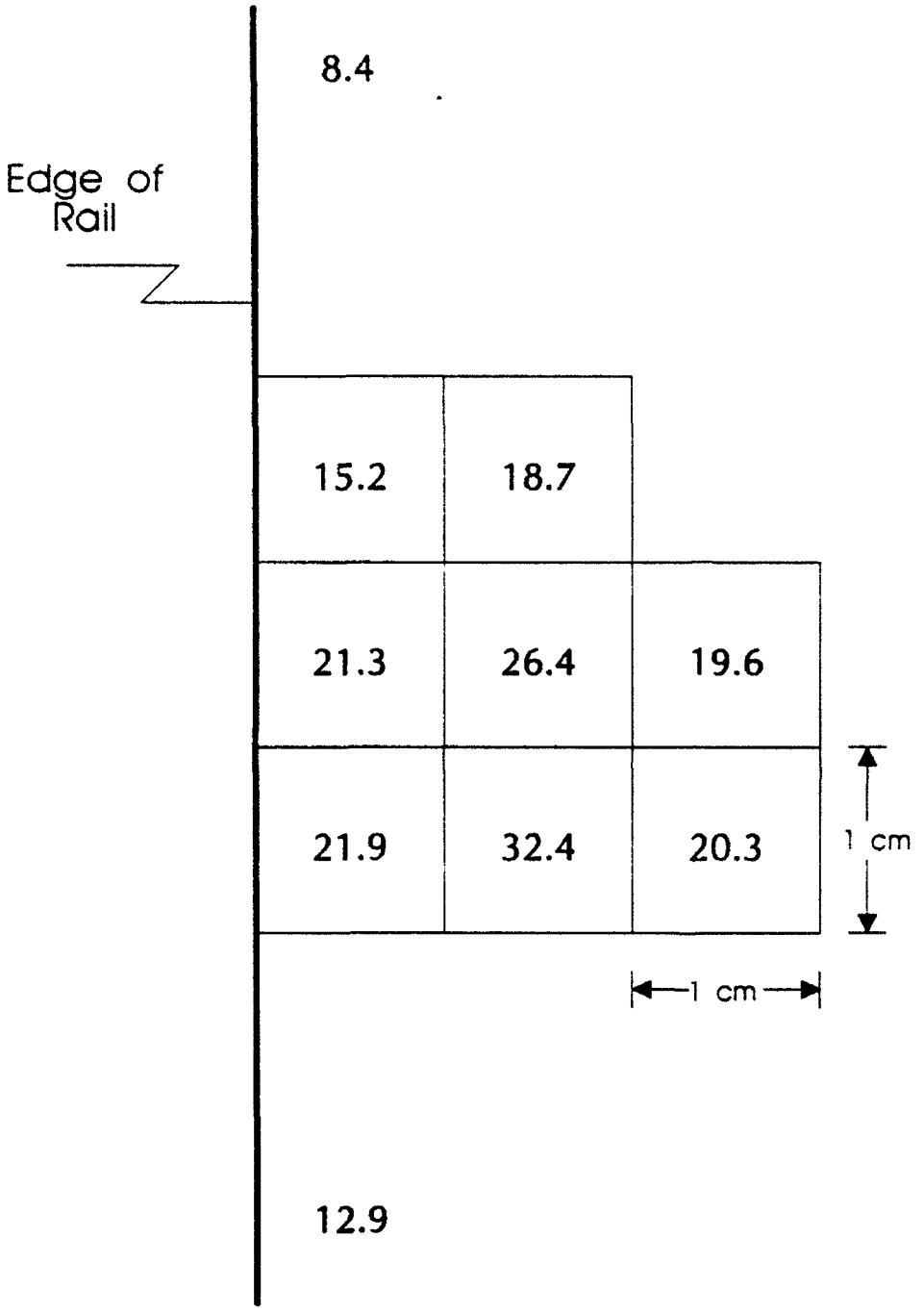

Figure 8: **TLD mapping of top of box around 'hot spot' - indicative of a crack in the shielding material allowing gamma-ray streaming. (Numbers represent dose rates in mrad/h).**

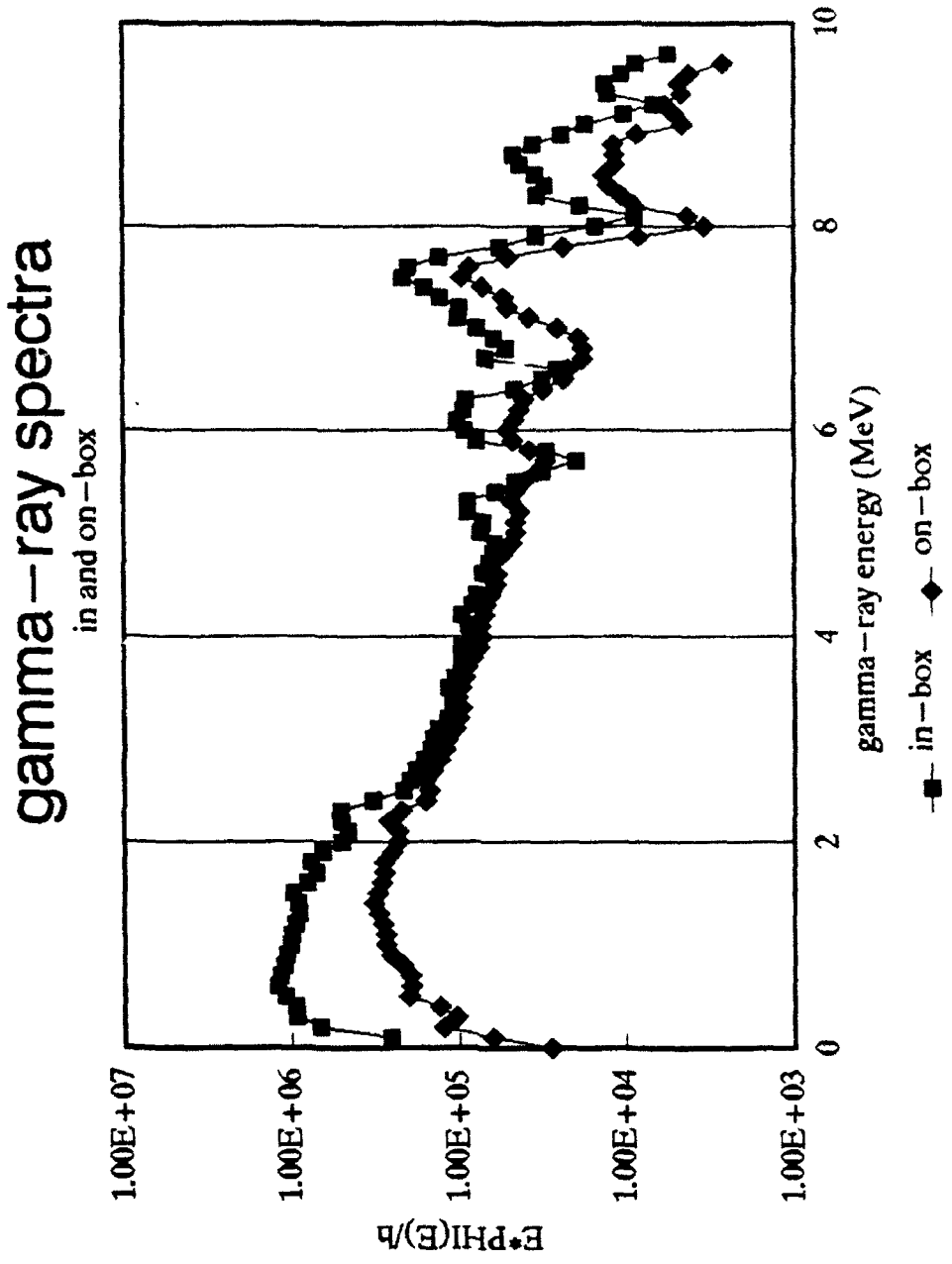

Figure 9: Comparison of BGO measured gamma-ray energy spectra in the box (1/20th power) and on top of the box (full power) both normalized to 1 hour.

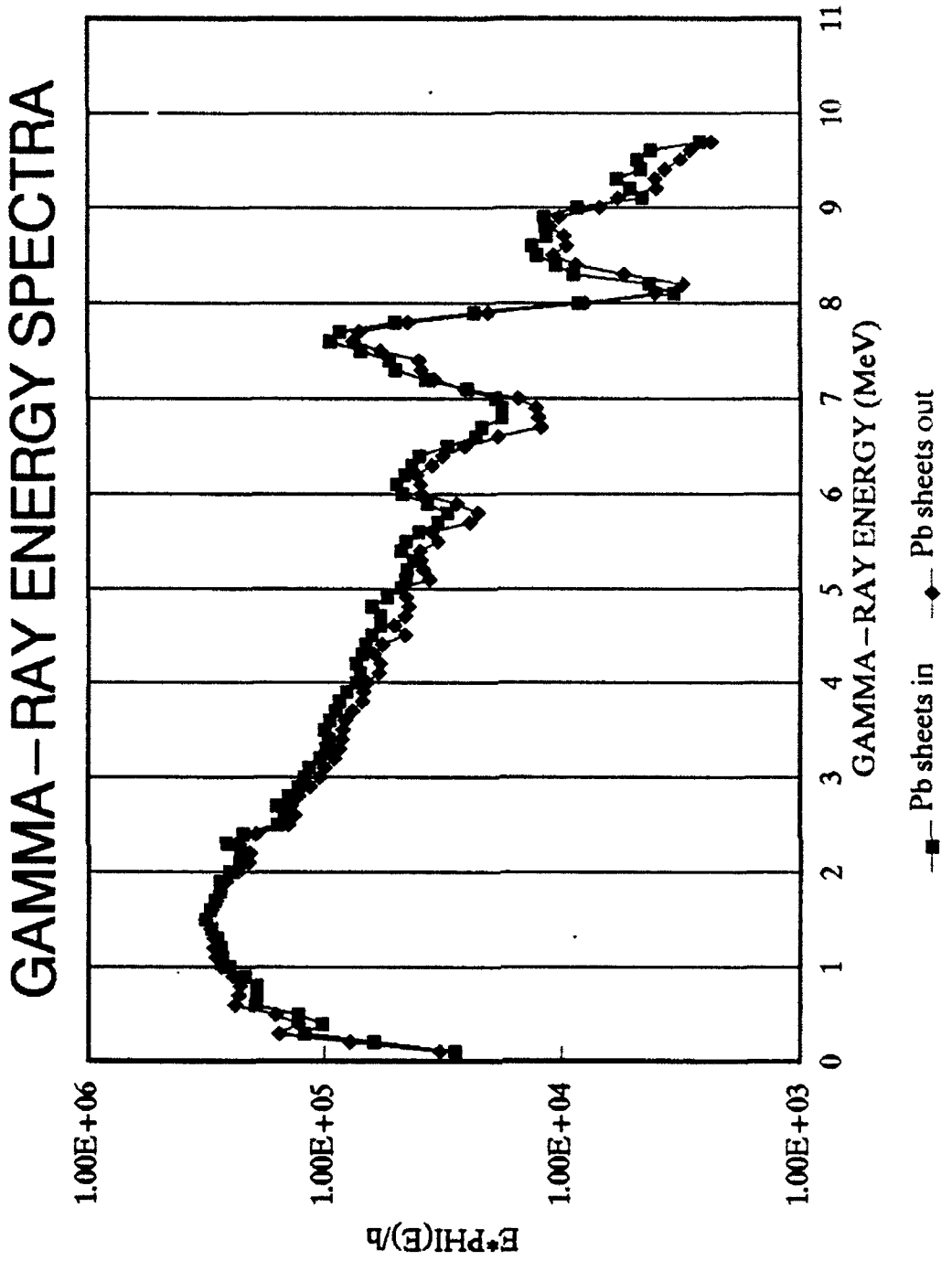

Figure 10: Comparison of BGO measured gamma-ray energy spectra on top of the box with Pb sheets in and out both at full reactor power for 1 hour exposure. Note that there is negligible effect.

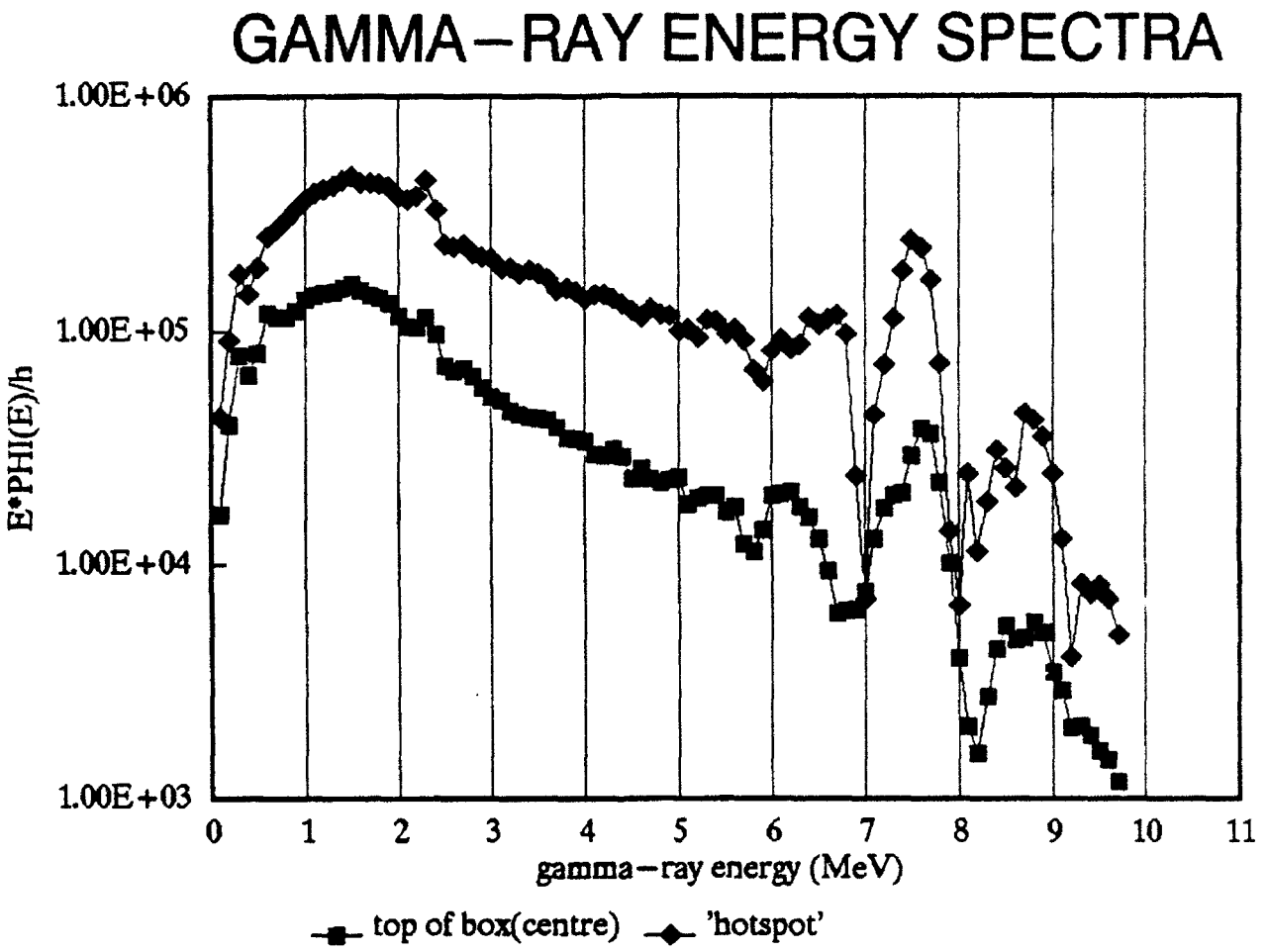

Figure 11: **Comparison of BGO measured gamma-ray energy spectra at centre of top of box and 'hot spot' both at full reactor power for 1 hour exposure.**

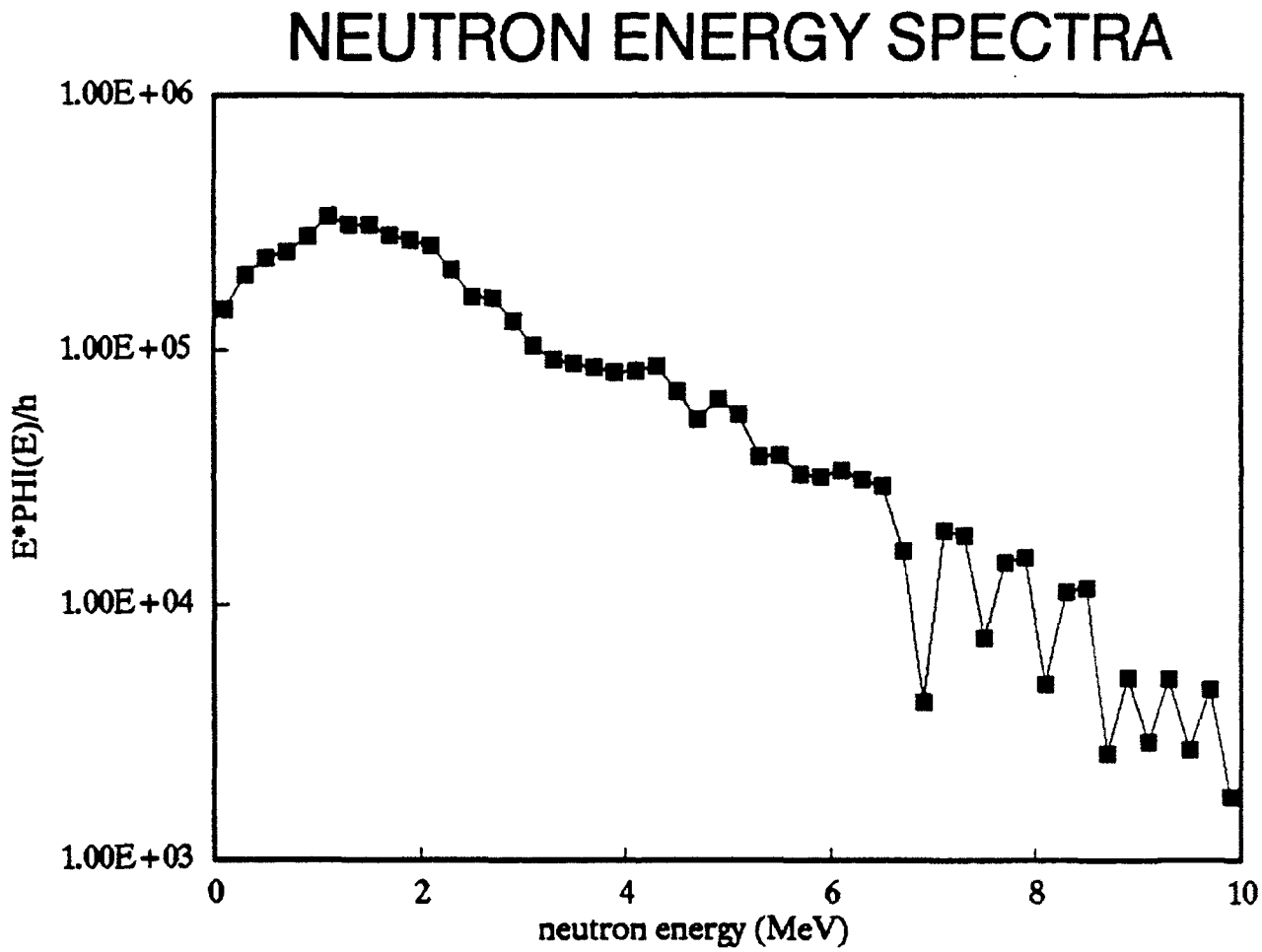

Figure 12: **NE213/BF, measured neutron energy spectrum on top of box at full reactor power, for 1 hour exposure.**

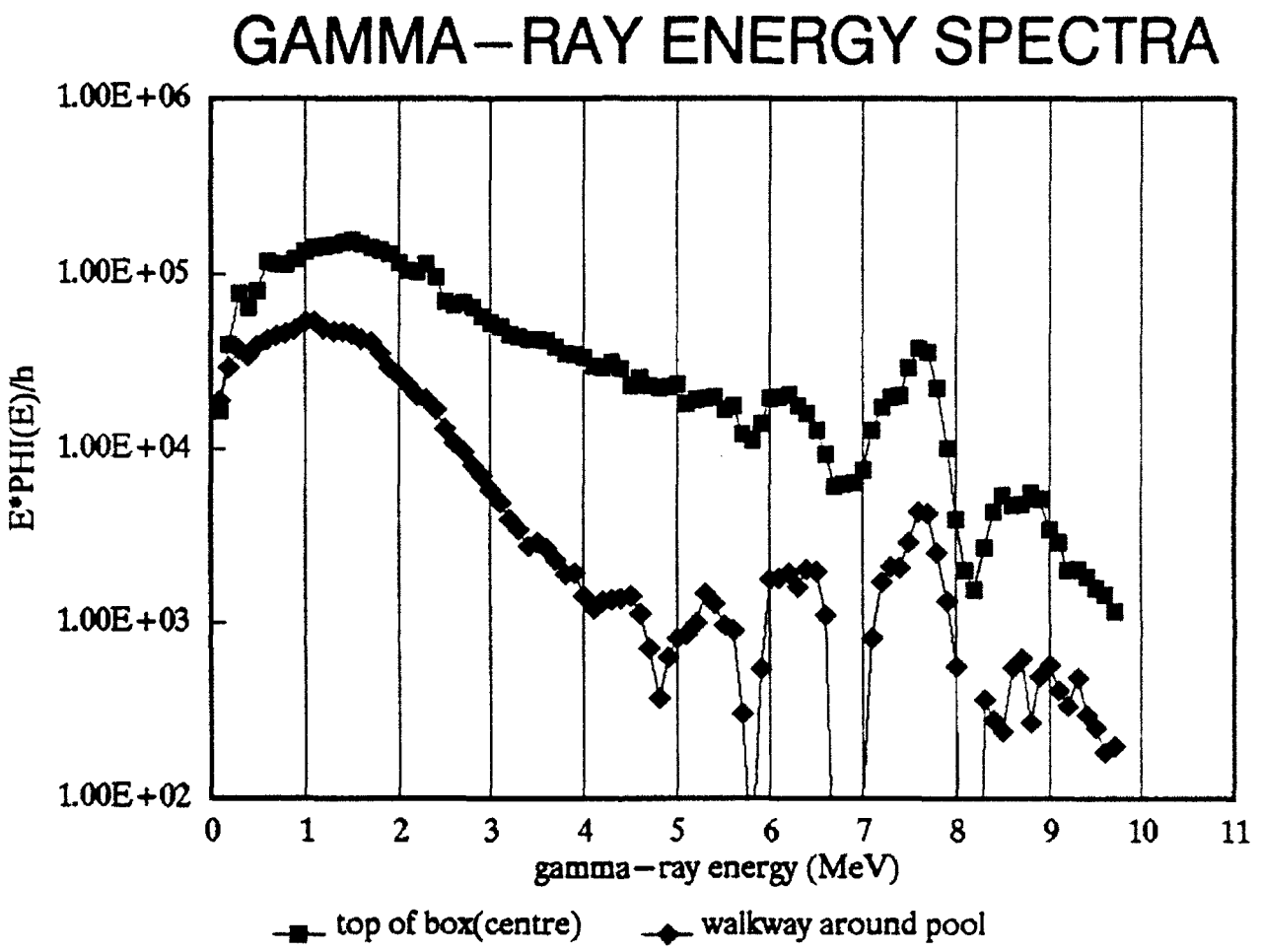

Figure 13: **Comparison of BGO-measured gamma-ray energy spectra on top of box and in walkway around pool. Both spectra are for full reactor power, and 1 hour exposure.**

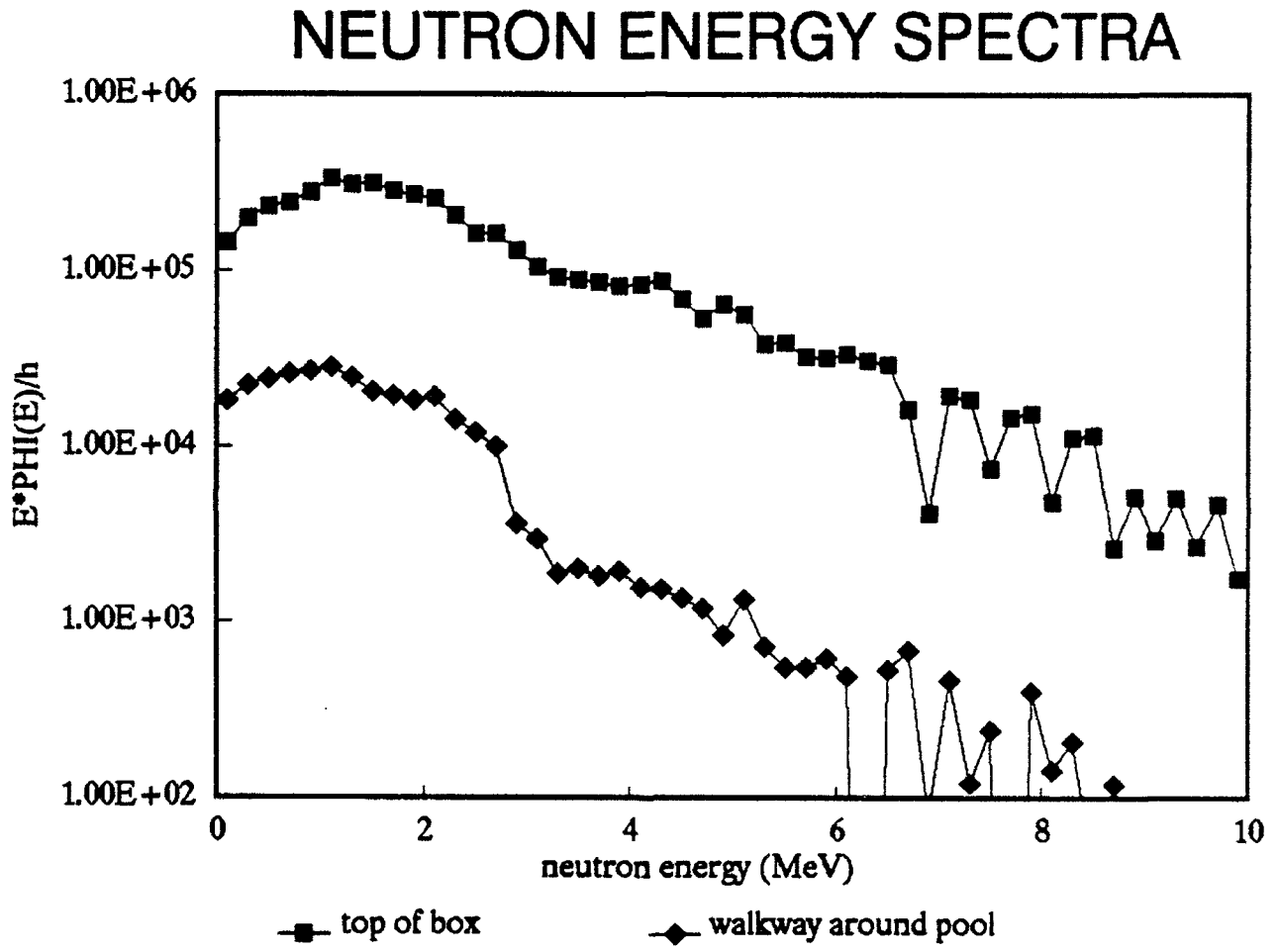

Figure 14: NE213/BF_3 measured neutron energy spectra for the same conditions as Figure 13.

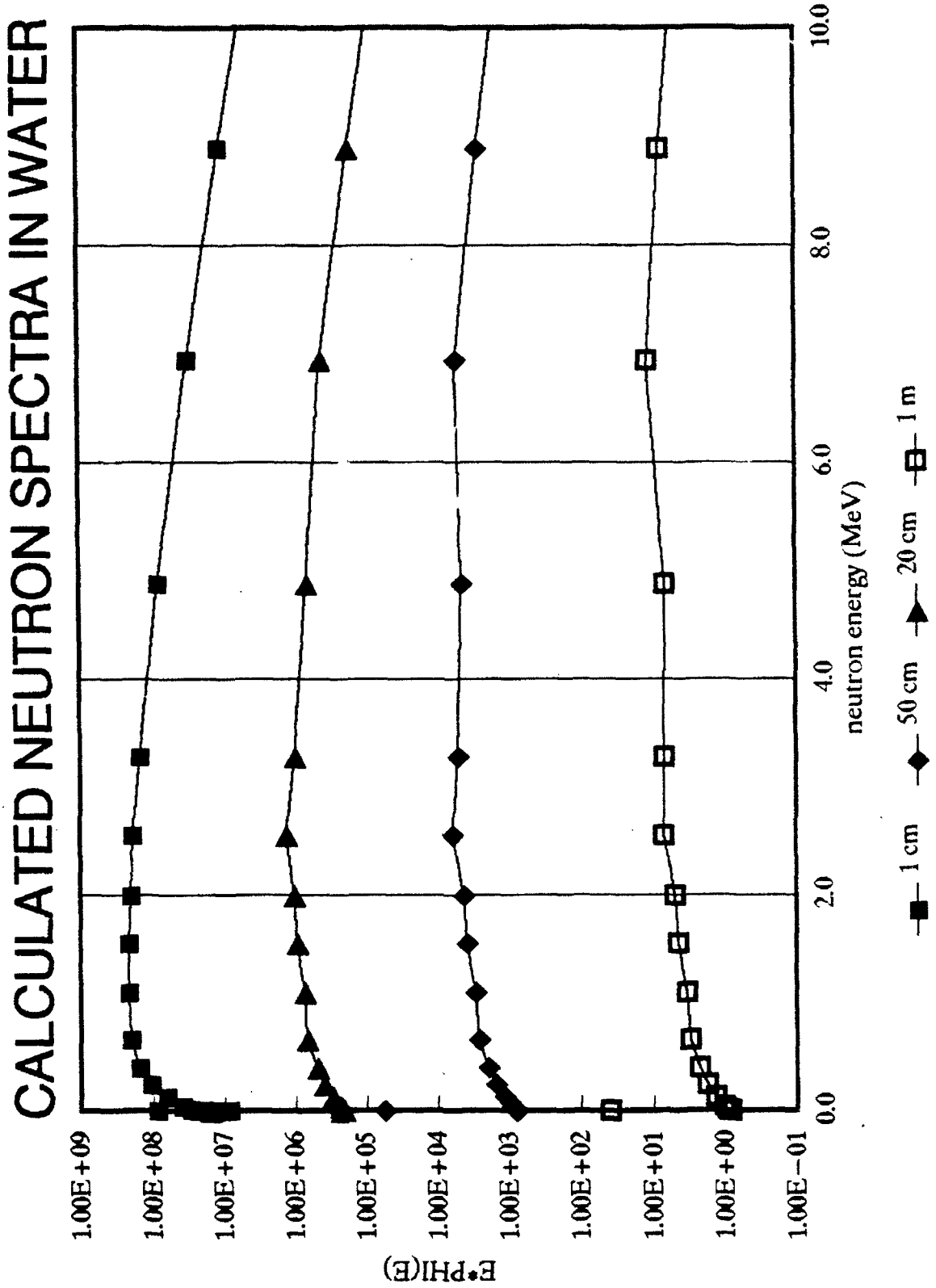

Figure 15: ANISN - calculated fission neutron energy spectra at various distances into a water moderator. Note that, even at 1m, a 1/E distribution is not valid below about 2 MeV.

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